UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 13, 2013

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION TO THE PRESSURE AND TEMPERATURE LIMIT CURVES AND THE LOW TEMPERATURE OVERPRESSURE PROTECTION LIMITS (TAC NO. MF0424)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment No. 281 to Renewed Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated December 14, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12353A319), as supplemented by letters dated January 31, 2013, and August 13, 2013 (ADAMS Accession Nos. ML13032A312 and ML13232A235).

The amendment revises the TMI-1 technical specifications with new pressure-temperature (P-T) limit curves and low-temperature overpressure protection system requirements. The P-T limit curves were revised to provide new curves that are valid to 50.2 effective full-power years for TMI-1.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

John D. Hughey

John D. Hughey, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures:
1. Amendment No. 281 to DPR-50
2. Safety Evaluation

cc: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 281
Renewed License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission or NRC) has found that:

   A. The application for amendment by Exelon Generation Company, LLC (the licensee), dated December 14, 2012, as supplemented by additional letters dated January 31, 2013, and August 13, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Renewed Facility Operating License No. DPR-50 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 281, are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective immediately and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John G. Lamb, Chief (Acting)
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: December 13, 2013

ATTACHMENT TO LICENSE AMENDMENT NO. 281

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following page of the Renewed Facility Operating License with the revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
Page 4	Page 4

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
vii	vii
3-3	3-3
3-4	3-4
3-5	3-5
3-5a	3-5a
3-5b	3-5b
--	3-5c
--	3-5d
--	3-5e
--	3-5f
3-18d	3-18d
3-18e	3-18e
3-18f	3-18f
4-41	4-41

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 281 are hereby incorporated in the license. The Exelon Generation Company shall operate the facility in accordance with the Technical Specifications.

(3) Physical Protection

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822), and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans[1], submitted by letter dated May 17, 2006, is entitled: "Three Mile Island Nuclear Station Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 3." The set contains Safeguards Information protected under 10 CFR 73.21.

Exelon Generation Company shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Exelon Generation Company CSP was approved by License Amendment No. 275.

(4) Fire Protection

Exelon Generation Company shall implement and maintain in effect all provisions of the Fire Protection Program as described in the Updated FSAR for TMI-1.

Changes may be made to the Fire Protection Program without prior approval by the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided that interim compensate measures are implemented.

(5) The licensee shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical parameters and control points for these parameters;
b. Identification of the procedures used to measure the values of the critical parameters;
c. Identification of process sampling points;
d. Procedure for the recording and management of data;

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[1] The Training and Qualification Plan and Safeguards Contingency Plan are Appendices to the Security Plan.

<u>LIST OF FIGURES</u>

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicability

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objectives

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

To assure that reactor vessel integrity by maintaining the stress intensity factor as a result of operational plant heatup and cooldown conditions and inservice leak and hydro test conditions below values which may result in non-ductile failure.

Specification

3.1.2.1 For operations until 50.2 effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1, 3.1-2, and 3.1-3 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figures 3.1-1 and 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figures 3.1-1 and 3.1-2. When the core is critical, allowable combinations of pressure and temperature shall be to the right of the criticality limit curve shown on Figure 3.1-1.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-3. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-3.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 DELETED

3.1.2.5 DELETED

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes (Reference 1). These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-1 of the UFSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation (Reference 2). The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the Nil Ductility Transition Temperature (NDTT).

The heatup and cooldown rate limits in this specification are based on linear heatup and cooldown ramp rates which by analysis have been extended to accommodate 15°F step changes at any time with the appropriate soak (hold) times. Also, an additional temperature step change has been included in the analysis with no additional soak time to accommodate decay heat initiation at approximately 240°F indicated RCS temperature.

The unirradiated reference nil ductility temperature (RT_{NDT}) for all Linde 80 welds were determined in accordance with BAW-2308, Rev. 1-A and Rev. 2-A, and 10 CFR 50, Appendices G and H. For the beltline plate and forging materials, 10 CFR 50 Appendices G and H were used to calculate the unirradiated reference nil ductility temperature (RT_{NDT}). For other beltline region materials and other reactor coolant pressure boundary materials, the unirradiated impact properties were estimated using the methods described in BAW-10046A, Rev. 2.

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT_{NDT} with accumulated nuclear operations. The adjusted reference temperatures have been calculated as described in Reference No. 5.

The predicted RT_{NDT} was calculated using the respective predicted neutron fluence at 50.2 effective full power years of operation and the procedures defined in Regulatory Guide 1.99, Rev. 2, Section C.1.1 for the plate metals and for the limiting weld metals (WF-70, WF-8, and SA-1526).

Analyses of the activation detectors in the TMI-1 surveillance capsules have provided estimates of reactor vessel wall fast neutron fluxes for cycles 1 through 17. Extrapolation of reactor vessel fluxes and corresponding fluence accumulations, based on predicted fuel cycle design conditions through 50.2 effective full power years of operation are described in Reference 6 with effective full power years clarified in Reference 7.

Based on the predicted RT_{NDT} after 50.2 effective full power years of operation, the pressure/temperature limits of Figures 3.1-1, 3.1-2, and 3.1-3 have been established by AREVA calculation, Reference No. 8, in accordance with the requirements of 10 CFR 50, Appendix G. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046A, Rev. 2 and ASME Code Section XI, Appendix G, as modified by ASME Code Case N-640 and N-588. The protection against nonductile failure is provided by maintaining the coolant pressure below the upper limits of these pressure/temperature limit curves. The minimum temperature for core criticality is determined to satisfy the regulatory requirements of 10 CFR Part 50, Appendix G. This limit is shown on Figure 3.1-1.

The pressure/temperature limit curves in Figures 3.1-1, 3.1-2, and 3.1-3 have been established considering the following:

a. System pressure is measured in RCS "A" loop hot leg. RCS "A" is most conservative and bounds use of "B".

b. Maximum differential pressure between the point of system pressure measurement and the limiting reactor vessel region for the allowable operating pump combinations.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

<u>REFERENCES</u>

(1) UFSAR, Section 4.1.2.4 - "Cyclic Loads"

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) DELETED

(4) DELETED

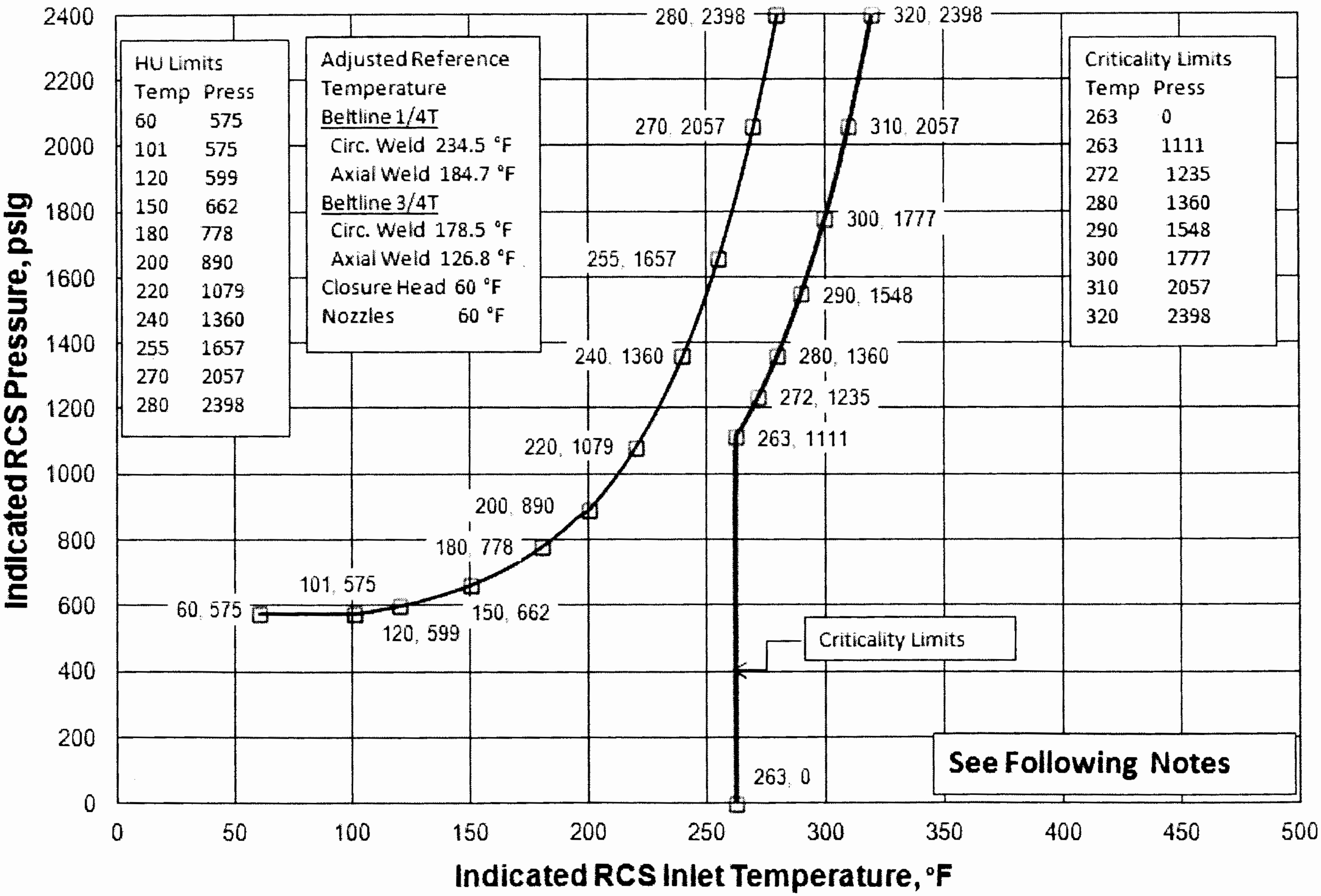
(5) AREVA Document No. 32-9035343, "ART Values for Three Mile Island at 60 Years."

(6) AREVA Document No. 86-9162844, "TMI-1 Cycles 16 and 17 Fluence Analysis Summary Report."

(7) AREVA Document No. 51-9035228, "TMI-1 Reactor Vessel Embrittlement Limits at 60 Years."

(8) AREVA Document No. 32-9177176, "TMI-1 Corrected P-T Limits at 50.2 EFPY."

Figure 3.1-1
Reactor Coolant System Heatup and Criticality Limitations
[Applicable through 50.2 EFPY]



Notes to Figure 3.1-1

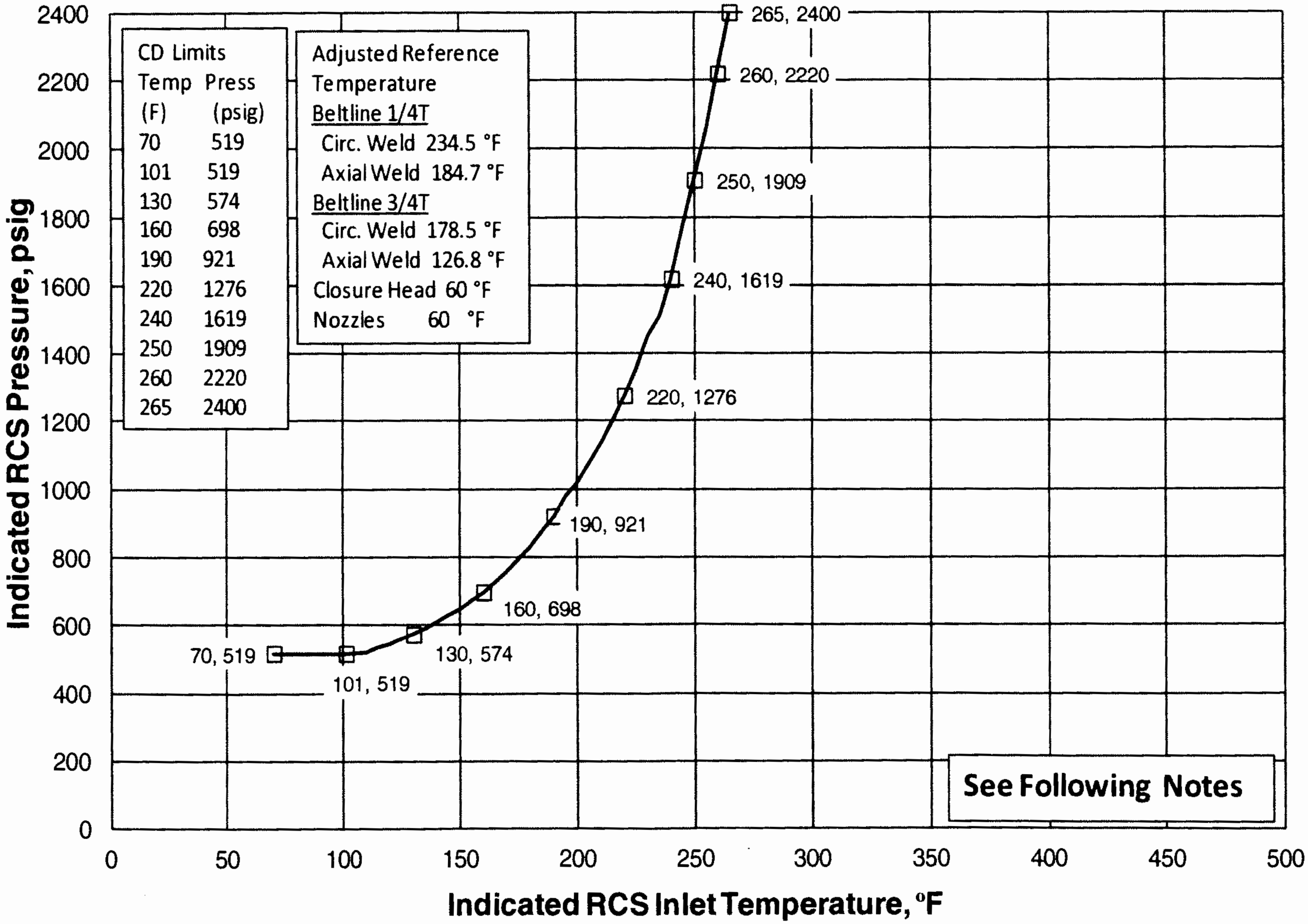
1 - Temperatures:
All Temperatures are the indicated values in the operating RC pump(s) Cold Leg.
Except:
When the DHRS is operating without any RC Pumps operating, then the indicated DHRS return temperature to the Reactor Vessel shall be used.

2 - Heatup:
50F/Hr or 15F/18 Min. Steps

3 - RC Pump Combinations for Heatups:
$T \leq 100$ No RC Pumps Operating
$100 < T \leq 199$ Any 1 or 2 Pump Combination (2/0, 0/2, 1/1)
$200 < T \leq 349$ Any Pump Combination except 2/2
$T \geq 350$ Any Pump Combination

4 - Criticality Limits:
When the core is critical, allowable combinations of pressure and temperature shall be to the right of the criticality limit curve.

Figure 3.1-2
Reactor Coolant System Cooldown Limitations
[Applicable through 50.2 EFPY]



Amendment ~~25, 134, 167, 176, 208, 234,~~ 281

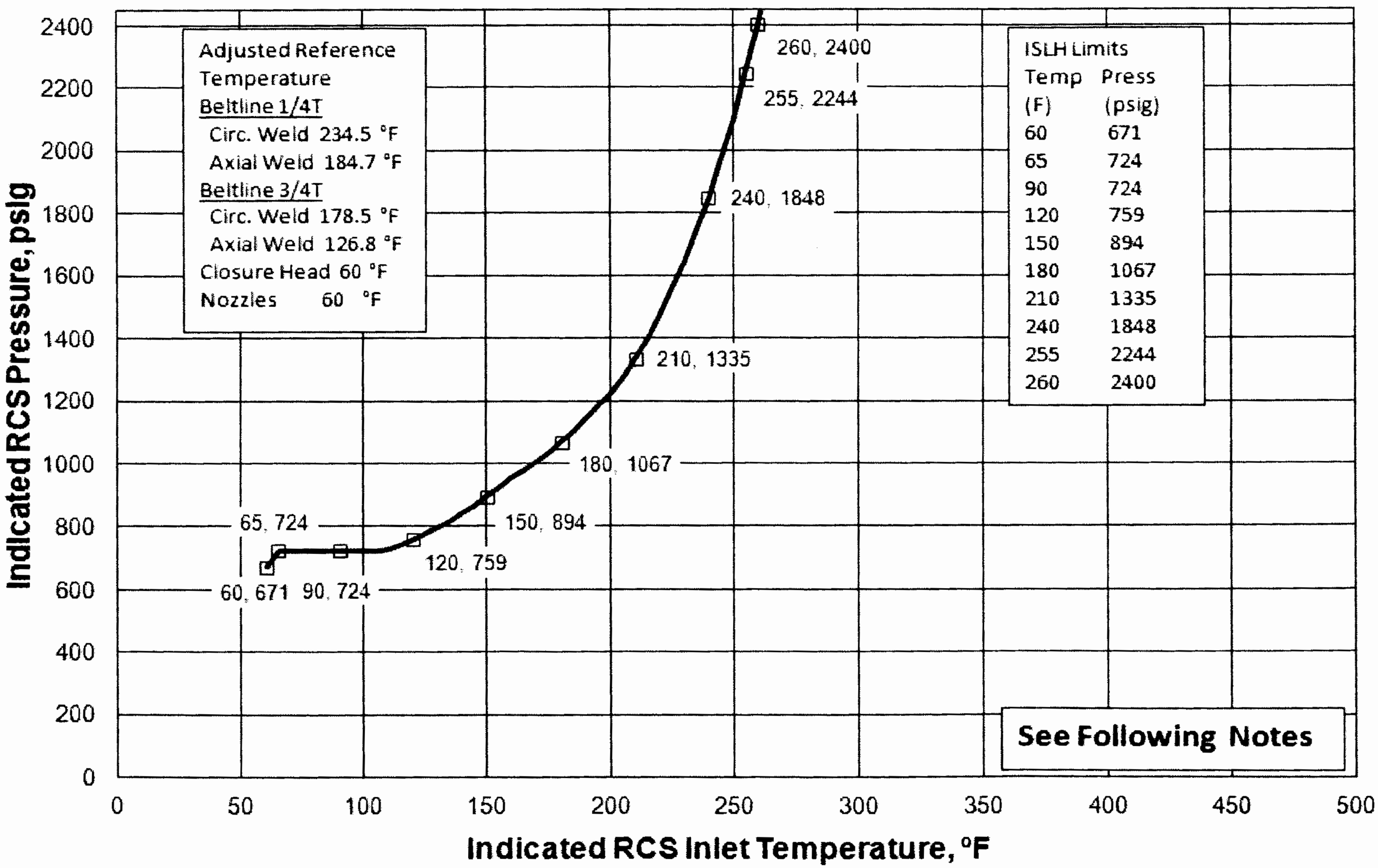
Notes to Figure 3.1-2

1 - Temperatures:
All Temperatures are the indicated values in the operating RC pump(s) Cold Leg.
Except:
When the DHRS is operating without any RC Pumps operating, then the indicated DHRS return temperature to the Reactor Vessel shall be used.

2 - Cooldown:
$T > 255F$ 100F/Hr or 15F/9 Min. Steps
$T \leq 255F$ 30F/Hr or 15F/30 Min. Steps

3 - RC Pump Combinations for Cooldowns:
$T \leq 100$ No RC Pumps Operating
$100 < T \leq 199$ Any 1 or 2 Pump Combination (2/0, 0/2, 1/1)
$200 < T \leq 349$ Any Pump Combination except 2/2
$T \geq 350$ Any Pump Combination

Figure 3.1-3
Reactor Coolant Inservice Leak Hydrostatic Test
[Applicable through 50.2 EFPY]



Notes to Figure 3.1-3

1 - Temperatures:
All Temperatures are the indicated values in the operating RC pump(s) Cold Leg.
Except:
When the DHRS is operating without any RC Pumps operating, then the indicated DHRS return temperature to the Reactor Vessel shall be used.

2 - Heatup:
50F/Hr or 15F/18 Min. Steps

3 - Cooldown:
$T > 255F$ 100F/Hr or 15F/9 Min. Steps
$T \leq 255F$ 30F/Hr or 15F/30 Min. Steps

4 - RC Pump Combinations for Heatups / Cooldowns:
$T \leq 100$ No RC Pumps Operating
$100 < T \leq 199$ Any 1 or 2 Pump Combination (2/0, 0/2, 1/1)
$200 < T \leq 349$ Any Pump Combination except 2/2
$T \geq 350$ Any Pump Combination

3.1.12 Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)

Applicability

Applies to the settings, and conditions for isolation of the PORV.

Objective

To prevent the possibility of inadvertently overpressurizing or depressurizing the Reactor Coolant System.

Specification

3.1.12.1 LTOP Protection

If the reactor vessel head is installed and indicated RCS temperature is ≤ 313°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed, and

b. Pressurizer level is maintained ≤ 100 inches. If pressurizer level is > 100 inches, restore level to ≤ 100 inches within 1 hour.

3.1.12.2 The PORV settings shall be as follows:

a. Low Temperature Overpressure Protection Setpoint

1. When indicated RCS temperature is ≤ 313°F, the LTOP system shall be operable as defined in Specification 3.1.12.1 and

2. The PORV will have a maximum lift setpoint of 592 psig.

With the PORV setpoint above the maximum value, within 8 hours either:

1. restore the setpoint below the maximum value, or

2. verify pressurizer level is ≤ 100 inches indicated and satisfy the requirements of Technical Specification 3.1.12.3 allowing the PORV to be taken out of service.

b. Unless the Low Temperature Overpressure Protection Setpoint is in effect, the PORV lift setpoint will be a minimum of 2425 psig.

With the PORV setpoint below the minimum value, within 8 hours either:

1. restore the setpoint above the minimum value, or

2. close the associated block valve, or

3. close the PORV, and remove power from PORV

4. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.1.12.3 When the indicated RCS temperature is below 313°F the PORV shall not be taken out of service, nor shall it be isolated from the system unless one of the following is in effect:

a. High Pressure Injection Pump breakers are racked out.
b. MU-V16A/B/C/D are closed with their breakers open, and MU-V217 is closed.
c. Head of the Reactor Vessel is removed.

3.1.12.4 The PORV Block Valve shall be OPERABLE during HOT STANDBY, STARTUP, and POWER OPERATION:

a. With the PORV Block Valve inoperable, within 1 hour either:

1. restore the PORV Block Valve to OPERABLE status or

2. close the PORV (verify closed) and remove power from the PORV

3. otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the PORV block valve inoperable, restore the inoperable valve to OPERABLE status prior to startup from the next COLD SHUTDOWN unless the COLD SHUTDOWN occurs within 90 Effective Full Power Days (EFPD) of the end of the fuel cycle. If a COLD SHUTDOWN occurs within this 90 day period, restore the inoperable valve to OPERABLE status prior to startup for the next fuel cycle.

<u>Bases</u>

If the PORV is removed from service while the RCS is below 313°F, sufficient measures are incorporated to prevent severe overpressurization by either eliminating the high pressure sources or flowpaths or assuring that the RCS is open to atmosphere.

The PORV setpoints are specified with tolerances assumed in the bases for Technical Specification 3.1.2. Above 313°F, the PORV setpoint has been chosen to limit the potential for inadvertent discharge or cycling of the PORV. Other action such as removing the power to the PORV has the same effect as raising the setpoint which also satisfies this requirement. There is no upper limit on this setpoint as the Pressurizer Safety Valves (T.S. 3.1.1.3) provide the required overpressure relief.

Below 313°F, the PORV setpoint is reduced to provide the required low temperature overpressure relief when high pressure sources and flowpaths are in service. There is no lower limit on the pressure actuation specified as lower setpoints also provide this same protection.

In both cases, the setting is specified to reflect the nominal value which allows for normal variations in the temperature setpoint while maintaining the tolerances assumed in the bases for T.S. 3.1.2. Either pressure actuation setpoint is acceptable above 313°F.

With RCS temperatures less than 313°F and the makeup pumps running, the high pressure injection valves are closed and pressurizer level is maintained less than 100 inches to allow time for action to prevent severe overpressurization in the event of any single failure.

The PORV block valve is required to be OPERABLE during the HOT STANDBY, STARTUP, and POWER OPERATION in order to provide isolation of the PORV discharge line to positively control potential RCS depressurization.

For protection from severe overpressurization during HPI testing, refer to Section 4.5.2.1.c.

4.5.2 EMERGENCY CORE COOLING SYSTEM

Applicability: Applies to periodic testing requirement for emergency core cooling systems.

Objective: To verify that the emergency core cooling systems are operable.

Specification

4.5.2.1 High Pressure Injection

a. At the frequency specified in the Surveillance Frequency Control Program and following maintenance or modification that affects system flow characteristics, system pumps and system high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable.

b. The test will be considered satisfactory if the valves (MU-V-14A/B & 16A/B/C/D) have completed their travel and the make-up pumps are running as evidenced by system flow. Minimum acceptable injection flow must be greater than or equal to 431 gpm per HPI pump when pump discharge pressure is 600 psig or greater (the pressure between the pump and flow limiting device) and when the RCS pressure is equal to or less than 600 psig.

c. Testing which requires HPI flow thru MU-V16A/B/C/D shall be conducted only under either of the following conditions:

1) Indicated RCS temperature shall be greater than 313°F. |
2) Head of the Reactor Vessel shall be removed.

4.5.2.2 Low Pressure Injection

a. At the frequency specified in the Surveillance Frequency Control Program and following maintenance or modification that affects system flow characteristics, system pumps and high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable. The auxiliaries required for low pressure injection are all included in the emergency loading sequence test specified in 4.5.1.

b. The test will be considered satisfactory if the decay heat pumps have been successfully started and the decay heat injection valves and the decay heat supply valves have completed their travel as evidenced by the control board component operating lights. Flow shall be verified to be equal to or greater than the flow assumed in the Safety Analysis for the single corresponding RCS pressure used in the test.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 281 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-50

REVISION TO THE PRESSURE AND TEMPERATURE LIMIT CURVES

AND THE LOW-TEMPERATURE OVERPRESSURE PROTECTION LIMITS

EXELON GENERATION COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

1.0 INTRODUCTION

By application dated December 14, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13232A214), as supplemented by letters dated January 31, 2013, and August 13, 2013 (ADAMS Accession Nos. ML13032A312 and ML13232A235), Exelon Generation Company (Exelon, or the licensee) requested changes to the technical specifications (TSs) for Three Mile Island Nuclear Station, Unit 1 (TMI-1). The proposed amendment would provide new pressure-temperature (P-T) limit curves and low-temperature overpressure protection (LTOP) system requirements. The licensee revised the P-T limit curves to provide new limits that are valid to 50.2 effective full-power years (EFPY) for TMI-1.

The licensee's submittal also requested an exemption related to the initial, unirradiated nil-ductility reference temperature (IRT_{NDT}) values. The exemption request seeks approval of an alternate initial nil-ductility reference temperature (RT_{NDT}) for Linde 80 weld materials per the Nuclear Regulatory Commission (NRC)-approved AREVA Topical Reports BAW-2308, Revision 1-A and Revision 2-A, "Initial RT_{NDT} of Linde 80 Weld Materials." The associated exemption was authorized and issued in NRC letter dated December 13, 2013 (ADAMS Accession No. ML13324A086).

The supplemental letters dated January 31, 2013, and August 13, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 19, 2013 (78 FR 16881).

Enclosure

2.0 REGULATORY EVALUATION

The NRC has established requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The NRC staff evaluates the P-T limits based on the following regulations and guidance:

- 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"
- 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"
- Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations"
- GL 92-01, Revision 1, "Reactor Vessel Structural Integrity"
- GL 92-01, Revision 1, Supplement 1
- Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" and
- Standard Review Plan (SRP), Branch Technical Position (BTP) 5-3, Revision 3, "Fracture Toughness Requirements"

Appendix G to 10 CFR Part 50 requires that P-T limits be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society for Mechanical Engineering (ASME), *Boiler and Pressure Vessel Code* (Code). Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves. GL 88-11 advised licensees that the NRC staff would use RG 1.99, Revision 2, to review P-T limits. RG 1.99, Revision 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation.

The GL 92-01, Revision 1, requested that licensees submit their reactor pressure vessel (RPV) materials property data for their plants to the NRC staff for review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

The SRP, BTP 5-3, Revision 3, "Fracture Toughness Requirements," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. ASME Code, Section XI, Appendix G, requires a safety factor of 2.0 on stress intensities resulting from pressure during normal and transient operating conditions, and a safety factor of 1.5 on these stress intensities for hydrostatic testing curves.

The flaw postulated in the ASME Code, Section XI, Appendix G, has a depth that is equal to ¼ of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Appendix G to Section XI of the ASME Code requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}) by evaluating material property changes due to neutron irradiation. The ART is defined as the sum of the initial RT_{NDT}, the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (σ) term. The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor (FF). The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The FF is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the neutron fluence and the calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the margin term.

AREVA TR [topical report] BAW-2308, Revision 1-A and Revision 2-A provide NRC-approved alternate initial RT_{NDT} and associated σ_I values for various heats of Linde 80 beltline weld materials for RPV integrity evaluation applications.

Section 50.60 of 10 CFR imposes fracture toughness and material surveillance program requirements, which are set forth in 10 CFR Part 50, Appendices G and H. Appendix G of 10 CFR Part 50 establishes fracture toughness requirements. In the "Definitions" section of Appendix G, paragraph G.II.D(ii) states, "For the reactor vessel beltline materials, RT_{NDT} must account for the effects of neutron radiation." In the "Fracture Toughness Requirements" section, paragraph G.IV.A states in part, "...the values of R_{NDT} and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of appendix H of this part." The effects of neutron radiation are determined, in part, by estimating the neutron fluence on the reactor vessel.

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence with respect to the General Design Criteria (GDC) contained in Appendix A of 10 CFR 50. In consideration of the guidance set forth in RG 1.190, GDC 14, 30, and 31 are applicable. GDC 14, "Reactor Coolant Pressure Boundary," requires the design, fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires among other things, that components comprising the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," pertains to the design of the reactor coolant pressure boundary, stating:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and

> postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

The construction permit for TMI-1 was issued by the Atomic Energy Commission (AEC) on May 18, 1968, and an operating license was issued on April 19, 1974. The plant design approval for the construction phase was based on the proposed GDC published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as "draft GDC"). The AEC published the final rule that added Appendix A, 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as "final GDC" or just "GDC"). Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. In accordance with an NRC staff requirement memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified During the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971, which includes TMI-1.

The TMI-1 Updated Final Safety Analysis Report (UFSAR), Section 1.4, provides an evaluation of the design bases of TMI-1 against the draft GDC. The UFSAR evaluation of the draft GDC, specifically, Criterion 9, "Reactor Coolant Pressure Boundary;" Criterion 16, "Monitoring Reactor Coolant Pressure Boundary;" Criterion 33, "Reactor Coolant Pressure Boundary Capability;" Criterion 34, "Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention;" Criterion 35, "Reactor Coolant Pressure Boundary Brittle Fracture Prevention;" and Criterion 36, "Reactor Coolant Pressure Boundary Surveillance;" reflect design requirements similar to those specified in the final GDCs specified in RG 1.190 as discussed above.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Change

The licensee proposes to revise the P-T limits in TMI, Unit 1 TS Section 3.1.2, "Pressurization Heatup and Cooldown Limitations." The proposed amendment will revise the reactor coolant system heatup, cooldown, and inservice leak hydrostatic test limitations for the Reactor Coolant System (RCS) to a maximum of 50.2 EFPY in accordance with 10 CFR 50, Appendix G. Further, the proposed amendment also revises TMI, Unit 1 TS Sections 3.1.12, "Pressurizer Power Operated Relief Valve (PORV), Block Valve, and Low Temperature Overpressure Protection (LTOP)," and TS 4.5.2, "Emergency Core Cooling System," for LTOP requirements to reflect the revised P-T limits of the reactor vessel.

3.2 Licensee's Evaluation for Alternate Initial RT_{NDT} and P-T limit curves

The proposed P-T limit curves in the licensee's letters dated December 14, 2012, and August 13, 2013, will replace the current P-T limit curves which were approved for 29 EFPY (License Amendment No. 308). The proposed curves address plant operations for Cycle 20 to the end of the extended period of operation (50.2 EFPY). The licensee updated the neutron fluence projections in 2011 to include the implementation of a Measurement Uncertainty Recapture (MUR) and confirmed the validity of the RPV embrittlement analyses in order to generate the new heatup and cooldown curves. The proposed curves use the most limiting ART values, determined with RG 1.99, Revision 2, and the alternate initial RT_{NDT} for Linde 80 weld materials per the NRC-approved TR BAW-2308, Revision 1-A and Revision 2-A.

The analysis supporting the proposed curves addressed three areas of the reactor coolant pressure boundary, the beltline shell region, the reactor coolant nozzles, and the closure head flange region. The beltline shell region and the reactor coolant nozzles are analyzed using 10 CFR Part 50, Appendix G, including the analytical methods and flaw acceptance criteria of TR BAW-10046A, Revision 2, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," and ASME Code Section XI, Appendix G. The closure head flange region is analyzed using generic limits that have been derived for reactor vessels designed by Babcock & Wilcox. The final curves presented include pressure corrections based on the location where the instrument for indicating pressure is located.

The P-T limits are based on the postulation of both axial and circumferential flaws in the most limiting axial and circumferential welds and the postulation of an axial flaw in the most limiting forging material of the reactor vessel. The licensee determined that the highest ART values at 50.2 EFPY are for the Linde 80 SA-1526 lower shell (LS) axial welds heat 299L44 ("axial flaw" orientation) and the lower nozzle belt (LNB) to upper shell (US) circumferential weld manufactured from Linde 80 WF-70 weld wire heat 72105 ("circumferential flaw" orientation). The licensee's ART determinations for these materials at the 1/4T and 3/4T locations were as follows:

Material	**Location**	**Neutron Fluence at Inside Surface (n/cm^2) (E>1 MeV)**	**ART (°F)**
LS axial weld SA-1526, heat 299L44	1/4T	1.177×10^{19}	184.7
LNB to US Circumferential Weld WF-70 Heat 72105	1/4T	1.836×10^{19}	234.5
LS axial weld SA-1526, heat 299L44	3/4T	1.177×10^{19}	126.8
LNB to US Circumferential Weld WF-70 Heat 72105	3/4T	1.836×10^{19}	178.5

The exemption from Appendix G to 10 CFR Part 50 requested the use of an alternate initial RT_{NDT} as described in TR BAW-2308, Revision 1 and Revision 2 for determining the adjusted RT_{NDT} of the Linde 80 weld materials present in the TMI-1 RPV beltline region. The exemption from 10 CFR 50.61 requests the use of an alternate methodology based upon direct fracture toughness testing per the Master Curve methodology, the 1997 and 2002 editions of American Society for Testing and Materials (ASTM) Standard Test Method E 1921 (ASTM E 1921), "Standard Test Method for Determination of Reference Temperature T_0, for Ferritic Steels in the Transition Range," and ASME Code Case N-629, "Use of Fracture Toughness Test Data to establish Reference Temperature for Pressure Retaining Materials, Section III, Division 1, Class

1." Pressurized thermal shock (PTS) reference temperature (RT_{PTS}) values projected to 60 calendar years were calculated and are summarized in Table 3-1 to Enclosure 1 of Attachment 3 to the December 14, 2012, submittal. Initial RT_{NDT} values and the corresponding σ_I values from TR BAW 2308, Revision 2-A and Revision 1-A were used for the Linde 80 welds.

3.3. Licensee's Evaluation for LTOP-limits

LTOP limits were based on the ASME Code, Section XI, Article G-2215, which requires that the LTOP system ensures that the maximum pressure from the limiting P-T curve is not exceeded when the 1/4T temperature is less than the ART + 50°F. During a cooldown, the licensee considered the coolant temperature is always less than (or equal to) the 1/4T temperature; therefore, it is conservative to use the coolant temperature as the LTOP enable set-point. However, during a heatup, the 1/4T temperature is always less than the corresponding coolant temperature. To support the development of the LTOP system limits, the temperature differences between the reactor coolant in the downcomer region and the 1/4T wall locations are determined for the maximum heatup rate transient.

The current LTOP enabling temperature and power-operated relief valve (PORV) maximum lift setpoint are 329°F and 552 psig, respectively; the proposed values are 313°F and 592 psig, based on the criteria specified in Appendix G of the ASME Code, Section XI, 1995 Edition through 1996 Addenda. Accordingly, the enabling LTOP temperature was determined using a value of RT_{NDT} for the 1/4T position of the circumferential weld projected to 50.2 EFPY of 235 °F, plus 50°F, plus 16°F to compensate for the maximum potential difference between the coolant temperature and the metal temperature during heatup. Factoring in instrument uncertainty (12°F), the LTOP will be enabled when the indicated coolant temperature decreases to 313°F.

For the PORV setpoint, the limiting condition is based on the allowable pressure at 60°F at 625 psig, based on the RPV closure head curve. With the maximum 13 psi correction for the indicating location, and the 20 psi PORV set-point uncertainty, the PORV would be set at 592 psig (i.e. 625-13-20 = 592 psig).

3.4. Licensee's Fluence Calculations

The licensee's supplemental letter dated January 31, 2013, described the fluence calculations supporting the license amendment request. As identified in the supplemental letter, the fluence methodology used by the licensee was that described in TR BAW-2241NP-A Rev2, "AREVA Topical Report: Fluence and Uncertainty Methodologies" (ADAMS Accession No. ML073310660).

3.5 NRC Staff Evaluation

3.5.1 Alternate Initial RT_{NDT}, RT_{PTS} and ART Values, and P-T Limit Curves

The NRC staff's safety evaluation (SE) for TR BAW-2308, Revision 1, required licensees to meet six conditions to use the methods of TR BAW-2308 Revision 1-A. The NRC staff's SE for TR BAW-2308, Revision 2, did not add any additional conditions to be resolved. One of the conditions was the submittal of a plant-specific exemption request.

The conditions are:

1. The initial reference temperature (IRT_{To}) and the uncertainty in IRT_{To} (σ_I) values given in Table 3 of the SE may be used by a licensee to define the initial heat-specific or generic properties of its facility's Linde 80 welds. For those Linde 80 weld wire heats for which heat-specific values are given, these values must be used when applying TR BAW-2308, Revision 1, if the heat-specific IRT_{To} value is more conservative than the generic "all heats" IRT_{To} value.

2. When the values from Table 3 of the SE are used by a licensee, the methodology of RG 1.99, Revision 2, may be used for the purpose of assessing the shift in initial properties due to irradiation (ΔRT_{NDT}), even though the RG 1.99, Revision 2, methodology is based upon Charpy V-notch (Cv) 30 ft-lb energy level shift data. However, based on the information in BAW 2308, Revision 1, a minimum chemistry factor of 167 °F must be applied when using initial properties given in Table 3 of this SE. A higher chemistry factor may be required if weld wire heat-specific chemical composition or Cv surveillance data indicate, via the methodology of RG 1.99, Revision 2, that a higher chemistry factor should apply.

3. When the values from Table 3 of the SE are used by a licensee, a value of σ_Δ (the uncertainty in ΔRT_{NDT}) of 28 °F must be used to determine the overall margin term, when the margin term per TR BAW-2308, Revision 1 is defined as:

$$\text{Margin} = 2\sqrt{(\sigma_I^2 + \sigma_\Delta^2)}$$

4. Any licensee who wants to utilize the methodology of TR BAW-2308, Revision 1, as outlined in items (1) through (3) above, must request an exemption, per 10 CFR 50.12, from the requirements of Appendix G to 10 CFR Part 50 and 10 CFR 50.61 to do so. As part of a licensee's exemption request, the NRC staff expects that the licensee will also submit information which demonstrates what values the licensee proposes to use for ΔRT_{NDT} and the margin term for each Linde 80 weld in its RPV through the end of its facility's current operating license.

5. The B&W Owner's Group (B&WOG) stated in their RAI response dated August 19, 2003, that fracture toughness data from one more heat of Linde 80 weld material (weld wire heat 61782) are to be obtained. The NRC staff expects the B&WOG to evaluate this data to determine whether or not the conclusions of TR BAW-2308, Revision 1, and this SE are nonconservative, and to communicate the B&WOG's conclusion to the NRC staff. Nonconservatism in TR BAW-2308, Revision 1, would be evident if: (1) the IRT_{To} value from the to-be-tested Linde 80 weld wire heat turns out to be higher than the generic IRT_{To} value approved in this SE, or (2) if the data from the to-be-tested Linde 80 weld wire heat results in an increase in the Linde 80 generic σ_I value.

6. Although the NRC staff concludes that there is reasonable assurance that the use of IRT_{To} values for Linde 80 weld materials, which were determined using the loading rate correction addressed in TR BAW-2308, Revision 1, is acceptable for the purpose of reactor vessel material property determination, the NRC staff expects that action will be

pursued within the appropriate consensus codes and standards organizations to address loading rate effects on a more generic basis (or determine that they do not need to be addressed) in the appropriate ASME Code Cases and/or ASTM Standard Test Methods. The staff requests that the B&WOG revise the recommended values in TR BAW-2308, Revision 1, in accordance with Table 3. When consensus codes and standards organizations address loading rate effects on a more generic basis, the NRC staff also expects that the B&WOG will re-evaluate TR BAW-2308, Revision 1, to determine whether or not revision of the TR is warranted.

The NRC staff's evaluation of the licensee's compliance with the conditions is as follows:

Condition # 1 was met because the licensee used the generic IRT_{To} and σ_I values for heats 8T1762 and 8T1554, for which no heat specific values are available.

Condition #2 was met since the licensee used the RG 1.99, Revision 2 method to determine the shift in the initial properties, and used a chemistry factor greater than 167 °F for both limiting materials, as verified by the staff's confirmatory calculation of the licensee's ART values.

Condition #3 was met since the NRC staff confirmatory calculation verified the margin terms provided in Table 3-1 of Enclosure 3 to the submittal dated December 14, 2012, which used a σ_Δ term of 28 °F and the σ_I value from Table 3 of the NRC staff SE for TR BAW-2308, Revision 1-A, for the Linde 80 weld wire heat 8T1762.

Condition #4 was met since the licensee requested an exemption, per 10 CFR 50.12, from the requirements of Appendix G to 10 CFR Part 50 and 10 CFR 50.61 and also submitted information that demonstrated what values the licensee proposes to use for ΔRT_{NDT} and the margin term for each Linde 80 weld in its RPV through the end of its facility's current operating license. The associated exemption was authorized and issued in NRC letter dated December 13, 2013 (ADAMS Accession No. ML13324A086).

Conditions 5 and 6 were resolved by TR BAW-2308, Revision 2, as documented in the final SE of that report.

Based on the above, the NRC staff finds that the licensee has met all six conditions for the use of the methodology of TR BAW-2308, Revision 1, for the determination of the initial material reference temperature. In the exemption required by Condition #4, the NRC staff found that the licensee will meet the intent of 10 CFR Part 50, Appendix G and 10 CFR 50.61. In addition, the licensee has used the revised initial RT_{NDT} and σ_I values from TR BAW-2308, Revision 2, for TMI-1 beltline materials containing Linde 80 weld wire heats 72105 and 299L44.

To determine the ART values of the beltline weld materials, the methodology of TR BAW-2308, Revisions 1-A and 2-A, was used to determine the initial, unirradiated RT_{NDT} of the controlling beltline materials. The licensee then used RG 1.99, Revision 2, to predict the shift in the material reference temperature (ΔRT_{NDT}). A margin term was then added in accordance with RG 1.99, Revision 2, except that the portion of the margin related to uncertainty in the initial reference temperature σi is determined using TR BAW-2308, Revision 1-A and Revision 2-A.

TR BAW-2308, Revision 1-A, provides an alternate method for determining adjusted RT_{NDT} and margins of the Linde 80 weld materials present in the beltline region of the TMI-1 RPV. By the SE dated June 27, 2007 (ADAMS Accession No. ML071160287), NRC staff approved the exemption request for the alternate material properties basis per 10 CFR 50.60(b) (which has been revised to 10 CFR 50.61(b)(4)).

In TR BAW-2308, Revision 2, the values of initial RT_{NDT} and initial margin terms for Linde 80 weld materials were changed slightly from the values approved in TR BAW-2308, Revision 1-A. The staff approved TR BAW-2308, Revision 2, in an SE dated March 24, 2008 (ADAMS Accession No. ML080770349).

Table 3-2 of Enclosure 3 to the submittal dated December 14, 2012, identifies the Linde 80 RPV beltline weld materials for TMI-1 and provided the material properties for these welds, including the initial reference temperature, ΔRT_{NDT}, for the 1/4T and 3/4T locations, and ART for the 1/4T and 3/4T locations. The values are projected to 60 years. The NRC staff performed independent calculations and confirmed that the controlling circumferential weld materials are the lower nozzle belt to US circumferential weld WF-70 with ART values of 234.5°F at the 1/4T RPV wall locations and 178.5°F at the 3/4T RPV wall location. The controlling 1/4T axial location is the lower shell axial weld SA-1526 with an ART of 184.7°F, and the controlling 3/4T axial location is the US weld axial weld WF-8 and lower shell plate C3307-1, both with ARTs of 126.8°F.

The licensee incorporated initial RT_{NDT} and σ_I values based on TR -2308, Revision 1-A and Revision 2-A, in projecting RT_{PTS} values applicable at 60 calendar years for the RPV beltline materials, including the Linde 80 beltline weld materials for TMI-1. The RT_{PTS} values for all beltline materials are summarized in Table 3-1 of Attachment 3 to the submittal dated December 14, 2012. The NRC staff performed an independent assessment of the RT_{PTS} values and concluded that all beltline materials for the TMI-1 RPV are projected to be below the PTS screening criteria at 60 years of operation. The NRC staff confirmed that the limiting beltline material is the lower nozzle belt to US circumferential weld WF-70, with a projected RT_{PTS} value of 263.8 °F. This is below the screening criterion of 300°F for circumferential welds. Therefore, the NRC staff concludes that the submittal is in accordance with the requirements of 10 CFR 50.61.

The licensee stated that the proposed P-T limit curves were based on the methodologies of Appendix G of Section XI of the ASME Code, 1995 Edition with the 1996 Addenda. The current and proposed P-T limit curves have also incorporated the provisions of ASME Code Case N-588 into Appendix G, which modified the methodology endorsed by the ASME Code regarding the postulation of circumferential weld flaws for the purpose of P-T limit generation and Code Case N-640, which allows the use of the ASME K_{Ic} curve instead of the K_{Ia} curve.

To assess the validity of the licensee's proposed curves, the NRC staff performed an independent assessment of the licensee's submittal. As part of the review, the NRC staff generated two requests for additional information (RAIs). First, the staff noted that in Section 4.6, "Reactor Coolant Temperature-Time Histories," of Attachment 4 to the original submittal dated December 14, 2012, the RPV coolant temperature-time histories used in the calculations of the proposed P-T limits are not consistent with Note 2 from Figure 3.1-2 in the proposed TS markup (Attachment 2 to the original submittal dated December 14, 2012). In RAI 1, the staff asked the licensee to explain this discrepancy, or revise the submittal to make the text consistent.

In the response dated August 13, 2013, the licensee noted that the cooldown ramp history in Note 2 of Figure 3.1-2 in the proposed TS markup is correct. A revision to Section 4.6 of Attachment 4 was included to correct the typographical error. The NRC staff noted that the proposed revision is consistent with the rest of the document and finds the issue raised in RAI 1 resolved.

The second RAI from the NRC staff noted that the properties for the nozzles used to construct the nozzle curves are not clearly documented. Therefore, the NRC staff requested that the licensee provide the inputs for evaluating the properties for the inlet and outlet nozzles, (i.e., chemistry and initial unirradiated RT_{NDT}, unirradiated Charpy USE), and neutron fluence at the end of the period of extended operation (PEO).

In the response dated August 13, 2013, the licensee stated that the inlet and outlet nozzle fluences are bounded by the 3.01×10^{16} n/cm^2, (E > 1.0 MeV) value reported in the TMI-1 License Renewal response to RAI 4.2.0.0-01 (ADAMS Accession No. ML082560178). Because the neutron fluence at the nozzles is less than 1×10^{17} n/cm^2, (E > 1.0 MeV), embrittlement in this region does not need to be considered for the PEO. The nickel contents are available on the certified material test records (CMTRs), but the initial RT_{NDT}, copper content, and Charpy USE were not measured for the nozzles during initial fabrication of the TMI-1 RPV and archive material was not maintained. Thus, certain material properties are not available for the inlet and outlet nozzle. The copper and nickel values are not relevant at this point because the chemistries are only required for the assessment of embrittlement, which does not need to be considered for the nozzles.

For the initial RT_{NDT} values, the TR BAW-10046A, Revision 2, topical report, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR Part 50, Appendix G," June 1986, provides an estimated value of 60°F for the TMI-1 RPV inlet and outlet nozzle forgings. For the USE values, the CMTRs include only +10°F Charpy USE data, which are not sufficient by itself to determine the USE per ASTM E185-82 (as directed by 10 CFR Part 50, Appendix G). The CMTRs show the average Charpy impact energy at +10°F is > 75 ft-lbs for each nozzle. Because the projected neutron fluence values for the nozzles demonstrated that embrittlement does not need to be considered (as discussed above), the nozzle forgings will have adequate USE at 50.2 EFPY.

The NRC staff has reviewed the response dated August 13, 2013, and finds the response acceptable because the projected 50.2 EFPY neutron fluence values for the inlet and outlet nozzle forgings will be well below the threshold for requiring embrittlement assessment; therefore, the copper chemistry and the ΔRT_{NDT} values are not necessary. Additionally, even if the ΔRT_{NDT} values were considered, judging from the distance between the heatup and cooldown P-T limits for nozzles and the corresponding composite bounding 32 EFPY P-T limits in TR BAW-10046A, Revision 2, it is unlikely that the small ΔRT_{NDT} for the nozzles would have any impact on the composite bounding P-T limits beyond 32 EFPY. Finally, the +10°F Charpy data from the CMTRs, as documented in Revision 20 of UFSAR for TMI-1, exceeds the 10 CFR 50, Appendix G USE requirements. With the acceptable response, the NRC staff considers the issues raised in RAI 2 to be resolved.

As part of the independent assessment, the NRC staff calculated the thermal stress intensity factors at the 1/4T location related to the maximum cooldown ramp history described in the August 13, 2013, revision to Section 4.6 of Attachment 4. The NRC staff based the thermal calculations on a report from Oak Ridge National Laboratory, ORNL/NRC/LTR-03/03, "Tabulation of Thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating Pressure-Temperature Curves per ASME Section XI – Appendix G," March 2003, (ADAMS Accession No. ML100840745). Combining the thermal stress intensity factors with the pressure-induced stress intensity factors, the NRC staff constructed an acceptable cooldown curve, which closely matched the proposed cooldown curve presented in the December 14, 2012, submittal. Both the NRC staff-calculated, and the licensee-proposed, cooldown curves bound the curves for the inlet and outlet nozzles, as well as the closure head curve.

Based upon the independent assessment, the NRC staff is satisfied that the licensee's proposed P-T limits are in accordance with Appendix G to Section XI of the ASME Code and satisfy the requirements in Paragraph IV.A.2 of Appendix G to 10 CFR Part 50.

3.5.2 LTOP Limits

The LTOP system, provided by the pressurizer PORVs, ensures RCS over-pressurization below certain temperatures would be prevented, thus maintaining reactor coolant pressure boundary integrity. The LTOP analysis yields limiting conditions for operation (LCOs) that constitute LTOP system alignments for the period of applicability.

The NRC staff reviewed the LTOP analysis using the guidance contained in Branch BTP 5-2, Rev 3, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." The NRC staff has confirmed that the enabling LTOP temperature was determined using a value of 235°F determined in 2006 during the License Renewal process, rather than the 216°F ART for the limiting Lower Nozzle Belt Forging to Upper Shell Circumferential Weld, 1/4T position, projected to 50.2 EFPY, presented in the technical basis document, ANP-3102, Revision 1 (Attachment 4 to the December 14, 2012, submittal).

For the PORV setpoint, the limiting condition is based on the allowable pressure at 60°F of 625 psig, based on the limiting RPV closure head curve. With the additional correction for the indicating location and the instrument uncertainty, the NRC staff notes that the PORV would provide adequate assurance of safe operation.

Given the use of the ASME Section XI, Article G-2215 methodology and additional conservative choice for the limiting ART value used by the licensee, the NRC staff accepts the analyses for the LTOP LCOs and concludes that the requested LTOP system limitations are acceptable.

3.5.3 Fluence Calculations

The NRC staff evaluated the fluence calculations to determine whether they adhere to the guidance contained in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The guidance provided in RG 1.190 indicates that the following attributes comprise an acceptable fluence calculation:

- A fluence calculation performed using an acceptable methodology;
- Analytic uncertainty analysis identifying possible sources of uncertainty;
- Benchmark comparison to approved results of a test facility;
- Plant-specific qualification by comparison to measured fluence values.

The fast neutron exposure parameters were determined for Exelon by AREVA, using the methods discussed in TR BAW-2241NP-A, Revision 2, "AREVA Topical Report: Fluence and Uncertainty Methodologies" (ADAMS Accession No. ML073310660). As noted by the SE titled "Revision 1 of Appendix G to TR BAW-2241(P), Revision 2, 'Fluence and Uncertainty Methodologies"' (ADAMS Accession No. ML061220721), dated April 28, 2006, the NRC staff determined that this methodology is generically acceptable for reference in licensing actions.

As described in TR BAW-2241NP-A, Revision 2, the neutron fluence was calculated using several methods. A solution to the Boltzmann transport equation is approximated using the two-dimensional discrete ordinates transport code. The licensee uses the cross-section library ENDF/B-VI. Approximations include a P3 Legrendre expansion for anisotropic scattering and a S8 order of angular quadrature. These approximations are in line with the P3 expansion and S8 quadrature suggested in RG 1.190. Space and energy dependent core power (neutron source) distributions and associated core parameters are treated on a fuel cycle specific basis. Three dimensional flux solutions are constructed using a synthesis of azimuthal, axial, and radial flux. Source distributions include cycle-dependent fuel assembly initial enrichments, burnups, and axial power distributions, which are used to develop spatial and energy dependent core source distributions that are averaged over each fuel cycle. This method accounts for source energy spectral effects by using an appropriate fission split for uranium and plutonium isotopes based on the initial enrichment and burnup history of each fuel assembly. The neutron calculations, as described above, are performed in a manner consistent with the guidance set forth in RG 1.190.

Also described in TR BAW-2241NP-A, Revision 2, an analytic uncertainty analysis was performed by combining the uncertainties associated with the individual components of the transport calculations in square-root-of-the-sum-of-the squares. The calculations were compared with the benchmark measurements from the Poolside Critical Assembly simulator at the Oak Ridge National Laboratory and vessel fluence benchmark problems provided in NUREG/CR-6115, "PWR [pressurized-water reactor] and BWR [boiling-water reactor] Pressure Vessel Fluence Calculation Benchmark Problems and Solutions," provided by Brookhaven National Laboratory. This constitutes acceptable test facilities.

Finally, TR BAW-2241NP-A, Revision 2, contains acceptable benchmarking for TMI-1, as it contains a database of PWR dosimetry benchmarking, and the TMI-1 unit geometry (B/W Reactor Vessel) is well represented within the database. The overall uncertainties (σ_C) of vessel fluence are 10.02 percent and 11.42 percent end of life vessel fluence. All reaction rates were calculated within 20-percent of measured values, as suggested in RG 1.190; therefore, the NRC staff determined that these uncertainties are acceptable.

3.6 Technical Conclusion

The NRC staff has reviewed the licensee's proposed changes and concludes that the proposed P-T limit curves and LTOP system requirements for TMI-1 satisfy the requirements in Appendix G to 10 CFR Part 50 and Appendix G to Section XI of the ASME Code. Hence, the proposed P-T limit curves and LTOP system requirements may be incorporated into the TMI-1 TSs and are valid through 50.2 EFPY. In addition, the NRC staff finds that the licensee has provided fluence calculations performed using an acceptable methodology, supported by analytic uncertainty analysis and a comparison to approved test facilities, along with comparison to plant-specific measured fluence values from surveillance capsules. Based on these considerations, the NRC staff concludes that the TMI-1 fluence calculations adhere to the guidance in RG 1.190, and that the neutron exposures reported in the licensee's submittal are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, and an inspection or surveillance requirement. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (78 FR 16881). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: P. Purtscher
M. Hardgrove

Date: December 13, 2013

December 13, 2013

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: REVISION TO THE PRESSURE AND TEMPERATURE LIMIT CURVES AND THE LOW TEMPERATURE OVERPRESSURE PROTECTION LIMITS (TAC NO. MF0424)

Dear Mr. Pacilio:

The Commission has issued the enclosed Amendment No. 281 to Renewed Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit 1 (TMI-1), in response to your application dated December 14, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12353A319), as supplemented by letters dated January 31, 2013, and August 13, 2013 (ADAMS Accession Nos. ML13032A312 and ML13232A235).

The amendment revises the TMI-1 technical specifications with new pressure-temperature (P-T) limit curves and low-temperature overpressure protection system requirements. The P-T limit curves were revised to provide new curves that are valid to 50.2 effective full-power years for TMI-1.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/ra/
John D. Hughey, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures:
1. Amendment No. 281 to DPR-50
2. Safety Evaluation

cc: Distribution via Listserv

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RidsAcrsAcnw_MailCTR Resource
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RidsNrrPMThreeMileIsland Resource
LPLI-2 R/F
PPurtscher, NRR
RidsDeEvib Resource
MHardgrove, NRR

ADAMS Accession No.: ML13325A023 * via memo ** via email

OFFICE	LPL1-2/PM	LPLI-2/LA	SRXB/BC	EVIB/BC	OGC (NLO)	LPL1-2/BC(A)	LPL1-2/PM
NAME	JHughey	ABaxter **	CJackson *	SRosenberg *	AGhosh	JLamb	JHughey
DATE	12/05/13	11/22/13	05/13/13	11/18 /13	11/27/13	12/06 /13	12/13 /13

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